Industry Perspectives on GSI-191 Closure

September 29, 2010

Ed Halpin
President and CEO
STP Nuclear Operating Company

STP Nuclear Operating Company

Mutual Agreement

- NRC Staff and Industry have done substantial work to drive this issue to closure
- Significant progress has been made in addressing GSI 191
- More probable small LOCA events should be addressed in the short-term
- Application of a risk informed approach for large LOCAs is appropriate based on extremely low risk

Proposed Small LOCA Resolution

- Target completion of testing by the end of 2011
- Licensees evaluate test results and commit to a resolution path by mid-year 2012
- Base any needed modifications on NRC accepted test results and acceptance criteria

Proposed Large LOCA Resolution

- Maintain all risk informed options available including an enhanced GDC 4
- Develop risk-informed methods and guidance specific to and appropriate for GSI-191
- Close the issue with commitments by mid-year 2012 for any additional actions, necessary modifications, and timelines on a plant specific basis

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Timeline Basis

Why is this timeline appropriate?

- Extremely low risk
- Industry has made significant progress in addressing GSI 191
- Allows completion of testing for small LOCA and application of PRA tools for large LOCA
- Test results may affect scope of additional modifications
- Adequate planning time minimizes radiological dose impacts
- Allows planning for aggregate impact of other regulatory issues

Why is a Holistic Risk Informed Approach Appropriate for Large LOCAs?

- Absolute versus reasonable certainty. What is needed to meet the adequate protection standard?
- Holistic risk informed approach is consistent with NRC principles
- All risk informed approaches remain available to licensees including 50.46(a), RG 1.174, and an enhanced GDC 4

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GSI 191 Closure Interactive Resolution

September 29, 2010

Peter P. Sena III
Senior Vice-President, FENOC Operations

FENOC

Beaver Valley is Substantially Complete

Item	Beaver Valley Unit 1	Beaver Valley Unit 2
Strainer	20 fold increase	20 fold increase
Piping Insulation	700 lineal feet	1700 lineal feet
SG Insulation	RMI (installed concurrent with RSG in 2006)	Replaced with RMI in 2009
Buffer Replacement	Not required	Sodium Tetraborate

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Beaver Valley is Substantially Complete

Impact	Beaver Valley Unit 1	Beaver Valley Unit 2
Cost	$25 M	$36 M
Dose	37 Rem	76 Rem
Outage duration (insulation replacement)	Did not impact critical path	Extended 2009 scheduled outage duration by 7 days

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Remaining Items for GSI-191 Closure

- Obtain final NRC approval of formal RAI responses
- Implement remaining outage modifications
- Resolve in-vessel effects

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FENOC Made Conservative Decisions Based on Deterministic Approach

- Our decisions were based on circumstances specific to our situation

- Risk-informed opportunity
  - Unit 2 SG insulation replacement (22 Rem)

- General Design Criteria 4 opportunity
  - Unit 2 SG insulation replacement (22 Rem)
  - Unit 1 Rx Nozzle insulation (12 Rem)
  - Unit 1 & 2 Piping insulation (10 Rem)
  - Unit 1 RCS primary piping insulation (6 Rem)

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Deterministic Approach - Leaves Limited Operational Margin

- Bounding analysis, levels of conservatism applied leave limited operating margin
- Strainer head loss results do not easily support extrapolation to higher debris loads
- Application of risk informed guidance could benefit operating margin

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Unresolved In-Vessel Effects Cause Ongoing Uncertainty

- Fuel may become limiting
- Potential for additional modifications or reduction in operating margin
- Application of risk informed guidance could benefit margin

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Summary

- Beaver Valley is substantially complete
- Our decisions were conservative, based on a deterministic approach and were specific to Beaver Valley
- Risk informed guidance could benefit operating margin
- Unresolved in-vessel effects causes ongoing uncertainty

FENOC

GSI –191
PWROG Resolution Efforts

September 29, 2010

Amir Shahkarami
Chairman PWROG Executive Committee
Senior VP Exelon, Site VP Braidwood Exelon Nuclear

PWROG
PWR Owners Group

Summary of Remaining PWROG Actions

- Zone of Influence Testing
- Long Term Core Cooling
- Method of Closure

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Zone of Influence Testing

- **Provided for both**
  - Deterministic approach (<12 inch breaks) and
  - Holistic risk approach (≥12 inch breaks) including GDC 4 and other potential mitigations
- **Staff was in agreement with the previous work on all but two issues:**
  - Scaling
    - Pipe break
    - Large components
  - Determination of Zone of Influence
    - Calculation of isobars
- **Schedule to complete testing and report: 12/2011**

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Long-Term Core Cooling Status

- **Cross test**
  - Conducted cross-test 9-09-10
  - Path forward - Options being evaluated:
    - Evaluating existing test results for applicability
    - Potential for further testing to support evaluation
    - Potential quantification of conservatisms in the testing process
    - Choose path forward with detailed resolution schedule by 10/4/10
- **Single train flow – dP**
  - Conducted tests at various participate-to-fiber (p:f) ratios
    - A 5:1 (p:f) test remains to be executed.

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Method of Closure

- **Two approaches should be used**
  - Closure for breaks smaller than the 12 inch pipe
  - Closure for breaks greater than or equal to 12 inch pipe

- **Timing of closure**
  - Smaller breaks (< 12 inch) addressed on shorter term schedule
  - Larger breaks (≥ 12 inch) addressed in manner that risk-informs schedule, methods and actions

PWROG
PWR Owners Group

Closure Method for Larger Breaks

- **Strong technical basis for application of GDC-4**
  - **Rigor in determination of lines and inspections**
    - Methodology typically used for 12-inch diameter and larger piping
    - Break probability is low, based on material properties
  - **PWSCC - inspection /mitigation**
    - Limited susceptible weld locations
    - Material Reliability Program (MRP) 139 –Primary System Piping Butt Weld Inspection and Evaluation Guidelines
    - Issued under NEI 03-08 and mandatory Implementation Schedule, Examination Requirements and Examination schedule
  - **Leak Monitoring**
    - LBB critical flaw size is a factor of 20 smaller on leakage vs. actual flaw size
    - PWROG Issued recommendation for implementation of RCS leak rate program guidelines under NEI 03-08 for any RCS leakage
      - Provides up to ~10x better leak detection than required for LBB

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PWR Owners Group

Capability to Maintain Defense-in-Depth

- **Safety Margins and Defense in Depth**
  - Significant safety margins are maintained in leakage detection for LBB critical flaw size vs. actual
  - Defense in Depth is maintained by combining LBB and ISI; application to GSI-191 does not decrease defense in depth
  - Double ended break of loop piping is much less probable than reactor vessel failure
  - EOP changes in response to NRC Bulletin 2003-01 provide additional defense in depth
  - Additional defense-in-depth measures and design modifications can be considered on plant specific basis
- **Changes in overall plant risk (CDF and LERF) are very small when LBB is applied to break $\geq$ 12 inch diameter**

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PWR Owners Group

Summary

- Agree with addressing small LOCA in short term
- Large LOCA resolution includes available holistic risk informed methods
- Proposed resolution timeline takes into account both risk and dose
- Complete analysis and testing prior to modifications
- Consider aggregate of NRC rulemaking

STP GSI 191 Cost and Dose

Previous SG insulation replacement experience

1RE09 43.8 REM
2RE09 73.0 REM
Total 116.8 REM

Dose to date (Rem)	Spent to date (Millions)	Estimated cost (Millions)	Estimated dose (Rem)
9.6	6.8	20 - 30	36 - 162

Acronyms

EOP	Emergency Operating Procedures
GDC-4	Appendix A to 10 CFR Part 50, General Design Criteria, Criterion 4, Environmental and dynamic effects design basis
ISI	In-service Inspection
LBB	Leak-Before-Break
LOCA	Loss-of-Coolant Accident
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Pressurized Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RMI	Reflective Metal Insulation
Rx	Reactor
SG	Steam Generator

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

GSI-191: Day 5,123

September 29, 2010

David Lochbaum

Director, Nuclear Safety Project

Union of Concerned Scientists

Acronym List

BI – barrier integrity
CRDM – control rod drive mechanism
ECCS – emergency core cooling systems
GSI – generic safety issue
IE – initiating event
LOCA – loss of coolant accident
MS – mitigating system
NOED – notice of enforcement discretion
NRC – Nuclear Regulatory Commission
ROP – reactor oversight process
UCS – Union of Concerned Scientists
URI – unresolved item

SECY-10-0113

NRC staff deserves credit for a paper that thoroughly explains the matter, including explanations for non-actions.

Example: UCS planned to advocate removing the in-vessel issue from GSI-191 until reading the staff's rationale on page 7.

GSI-191 Closure Options

Regardless of option(s) chosen, GSI-191 closure will take years

SECY page 4: '...none of the options below provide an "analysis only" option...'

Until the non-analytical actions are completed, reactors are operating at elevated risk

GSI-191 Closure Options

NRC staff recommends Options 1.b and 2 for GSI-191 close-out

UCS supports the staff's recommendation with one important caveat – the risks associated with unresolved GSI-191 issues must be formally factored into NRC's regulatory decision-making

GSI-191 Closure Options

The 1.b/2 approach allows the pending 50.46a rulemaking to address the leak before break application issue

UCS finds the NRC staff's evaluation of the worker radiation dose question to be sound and persuasive

GSI-191 Closure Options

UCS agrees with NRC staff that Option 3 is not really a closure option.

In essence, Option 3 says the public is protected from a large break LOCA, unless a large break LOCA occurs.

That's unacceptable public policy.

GSI-191 Reduces Risk

Threat that containment sump screens for ECCS pumps would get clogged by debris during a LOCA is reduced by enlarging screens and reducing amount of potential debris

Until resolved, GSI-191 represents elevated risks

Table 5-7 Results of Parametric Evaluations Regarding Potential for Blockage

	ID	SLOCA	MLOCA	LLOCA	ID	SLOCA	MLOCA	LLOCA	
	1	Likely+	Very Likely+	Very Likely	36	Very Likely+	Very Likely	Very Likely	
	2	Unlikely	Possible	Very Likely	37	Very Likely	Very Likely	Very Likely	
	3	Unlikely	Unlikely	Likely	38	Unlikely	Unlikely	Likely	
	4	Very Likely	Very Likely	Very Likely	39	Unlikely	Possible	Very Likely	
	5	Very Likely+	Very Likely+	Very Likely	40	Unlikely	Unlikely	Very Likely	
	6	Likely	Very Likely	Very Likely	41	Unlikely	Unlikely	Likely	
	7*	Unlikely	Unlikely	Unlikely	42	Likely+	Very Likely	Very Likely	Salem 2
	8	Very Likely	Very Likely	Very Likely	43	Unlikely	Unlikely	Very Likely	
	9	Very Likely	Very Likely	Very Likely	44	Unlikely	Unlikely	Very Likely	
San Onofre 2	10	Very Likely+	Very Likely+	Very Likely	45	Very Likely+	Very Likely+	Very Likely	Palisades
	11	Very Likely+	Very Likely+	Very Likely	46	Unlikely	Possible	Very Likely	
	12	Possible	Very Likely+	Very Likely	47	Very Likely	Very Likely	Very Likely	Indian Point 2
	13	Unlikely	Unlikely	Very Likely	48	Very Likely	Very Likely	Very Likely	
	14	Unlikely	Unlikely	Very Likely	49*	Unlikely	Unlikely	Unlikely	
Seabrook	15	Unlikely	Likely	Very Likely	50	Unlikely	Unlikely	Possible	
	16	Very Likely+	Very Likely+	Very Likely	51	Very Likely+	Very Likely+	Very Likely+	
	17	Very Likely	Very Likely	Very Likely	52	Unlikely	Unlikely	Likely	
	18*	Unlikely	Unlikely	Unlikely	53	Likely	Very Likely	Very Likely	
Indian Point 3	19	Very Likely	Very Likely	Very Likely	54	Likely+	Likely	Very Likely	
	20	Very Likely	Very Likely	Very Likely	55	Possible	Likely+	Very Likely	Millstone 2
	21	Unlikely	Possible	Likely	56	Unlikely	Unlikely	Very Likely	
HB Robinson	22	Very Likely+	Very Likely	Very Likely	57	Unlikely	Unlikely	Very Likely	
	23	Unlikely	Possible	Very Likely	58	Very Likely	Very Likely	Very Likely	Millstone 3
	24*	Unlikely	Unlikely	Unlikely	59	Very Likely	Very Likely	Very Likely	
	25	Possible+	Possible+	Very Likely	60	Unlikely	Likely	Very Likely	
	26	Very Likely	Very Likely	Very Likely	61	Unlikely	Unlikely	Likely	
	27	Likely+	Likely	Very Likely	62	Very Likely+	Very Likely+	Very Likely	San Onofre 3
Salem 1	28	Likely+	Very Likely	Very Likely	63	Very Likely	Very Likely	Very Likely	
	29*	Unlikely	Unlikely	Unlikely	64*	Unlikely	Unlikely	Unlikely	
	30	Possible+	Unlikely	Very Likely	65	Very Likely	Very Likely	Very Likely	
	31*	Unlikely	Unlikely	Unlikely	66*	Unlikely	Unlikely	Unlikely	
	32	Very Likely	Very Likely	Very Likely	67	Unlikely	Unlikely	Very Likely+	Davis-Besse
	33	Unlikely	Likely+	Very Likely	68	Unlikely	Unlikely	Very Likely	
	34	Unlikely	Unlikely	Very Likely+	69	Unlikely	Unlikely	Likely	
	35	Very Likely+	Very Likely+	Very Likely					

Source: NUREG/CR-6762 v1 dated 08/2002

Ignored Risk Factor

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ECCS - Operating
B 3.5.2

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system,

b. Rod ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 24 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

ECCS recirculation phase "Very Likely" to be lost during large LOCA at 37 reactors before GSI-191 fixes

Source: NUREG-1431, R1 12/2005

Ignored Risk Factor

BASES

APPLICABILITY (continued)

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

Without ECCS, reactor can only be "safely" operated for 72 hours, not 5,132 days

Source: NUREG-1431, R1 12/2005

Ignored Risk Factor

NRC staff justified operation of reactors while GSI-191 was resolved based on low probability that LOCA would occur

In isolation, justification may be appropriate; but reactors did not and will not operate with only unresolved GSI-191 safety issues

Ignored Risk Factor

While GSI-191 was unresolved, NRC issued Bulletin 2002-02 about increased probability of a LOCA due to CRDM nozzle cracking

NRC staff justified operation of reactors while Bulletin 2002-02 was resolved based on high reliability of ECCS systems

Ignored Risk Factor

ECCS impairment (GSI-191) was accepted based on low likelihood of LOCA

Increased LOCA likelihood (Bulletin 2002-02) was accepted based on unimpaired ECCS performance

NRC made both decisions in isolation

Ignored Risk Factor

UCS is not saying or implying that either of these NRC decisions was wrong or would have been different had all known risk factors been considered

BUT, we are saying that it is wrong to make risk decisions ignoring known risk factors

Risk-Informed Decision-Making

To properly reach risk-informed regulatory decisions, the risk of known but unresolved safety issues must be considered, at least qualitatively

Until resolved, GSI-191 risks along with risks from all open Bulletins, Generic Letters, NOEDs, URIs, etc must be considered

Risk-Informed Decision-Making

Example: A decision for a reactor with known Initiating Event problems might differ from that made for reactors with no known problems.

2Q/2010 ROP Performance Indicators Summary

Plants	IE 01	IE 03	IE 04	MS 05	MS 06	MS 07	MS 08	MS 09	MS 10	BI 01	BI 02	EP 01	EP 02	EP 03	OR 01	PR 01
Arkansas Nuclear 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Arkansas Nuclear 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Springfield 1	R	Y	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Springfield 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 1	G	G	G	G	I	I	I	I	I	G	G	G	G	G	G	G
Browns Ferry 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Risk-Informed Decision-Making

Many utilities use a design change checklist to review whether a proposed modification could adversely impact fire protection, environmental qualification, seismic design, and other safety factors.

Risk-Informed Decision-Making

NRC could employ a risk-informed decision checklist to review whether a proposed decision could affect or be affected by plant-specific unresolved safety issues (e.g., GSI-191, NEODs, inspection report URIs, greater-than-green ROP results, one-time exemptions, etc.)

Conclusion

UCS supports the NRC staff's recommendation that Options 1.b and 2 be used to closeout GSI-191

Regardless of options chosen, the risks associated with unresolved GSI-191 issues must be formally factored into NRC's regulatory decision-making

Back-up Slides

MATRIX OF LICENSEE MODIFICATIONS TO ADDRESS GENERIC SAFETY ISSUE 191

Source: NEI, 01/30/2007

Plant (Note: Asterisk indicates extension request approved or requested - See following table)	Original Strainer Area (Ft2)	Estimated Total size of Replace-ment Strainer (Ft2)	Strainer vendor	Strainer area shared between trains? (Yes/No)	Planned or actual quarter / year for strainer installation	pH Buffer change being pursued? (Yes/No)	If Yes, planned or actual quarter / year for pH Buffer change	Major (20% or more) insulation change out planned? (Yes/No)	If Yes, planned or actual quarter/year for completion of the insulation change	Major ECCS system mods (e.g., throttle valve/pump changes and cyclone separator removal) planned? (Yes/No)	If Yes, planned or actual quarter/year for completion of the changes	Any major containment spray system modifications planned? (Yes/No)	If Yes, planned or actual quarter/year for completion of the changes
Arkansas Nuclear One 1	179	TBD	CCI	Yes	1Q07	TBD	N/A	TBD	N/A	TBD	N/A	Ongoing	N/A
Arkansas Nuclear One 2	154	~4600	CCI	Yes	3Q06	TBD	N/A	No	N/A	TBD	N/A	Ongoing	N/A
Beaver Valley 1	130	3000	CCI	Yes	4Q07	No	N/A	No	N/A	Yes – throttle valve replacement	4Q07	Yes	4Q07
Beaver Valley 2*	150	3300	Enercon	Yes	4Q06	No	N/A	Yes	1Q08	Yes – throttle valve replacement	1Q08	Yes	1Q08
Braidwood 1	150 (total two sumps)	3000 per sump (2 sumps)	CCI	No	4Q07	No	N/A	Yes	4Q07	Yes - throttle valve trim replacement; separator testing	4Q07	No	N/A
Braidwood 2*	150 (total two sumps)	3000 per sump (2 sumps)	CCI	No	4Q06	No	N/A	No	N/A	Yes - throttle valve trim replacement; separator testing	2Q08	No	N/A
Byron 1*	150 (total two sumps)	3000 per sump (2 sumps)	CCI	No	3Q06	No	N/A	Yes	3Q06	Yes - throttle valve trim replacement; separator testing	2Q08	No	N/A
Byron 2	150 (total two sumps)	3000 per sump (2 sumps)	CCI	No	2Q07	No	N/A	No	N/A	Yes - throttle valve trim replacement; separator testing	2Q07	No	N/A
Callaway	~400 total (both trains)	~6600 total (both trains)	PCI	No	2Q07	No	n/a	No	N/A	No, not currently. Potential minor modification to cyclone separators	N/A	No	N/A
Calvert Cliffs 1*	102	6000	CCI	Yes	1Q07	No	N/A	No	N/A	TBD	N/A	No	N/A
Calvert Cliffs 2	102	6000	CCI	Yes	1Q08	No	N/A	No	N/A	TBD	N/A	No	N/A
Catawba 1*	135	2441	Enercon	Yes	2Q08	No	N/A	TBD	TBD	Yes - ECCS Injection Line Orifice replacements	4Q06	Under consideration	TBD
Catawba 2	135	2441	Enercon	Yes	4Q07	No	N/A	TBD	TBD	Yes - ECCS Injection Line Orifice replacements	4Q07	Under consideration	TBD
Comanche Peak 1	260 per sump	3947 per sump	PCI	No	1Q07	Yes	4Q07 or later	No	N/A	No	N/A	No	N/A
Comanche Peak 2	260 per sump	3947 per sump	PCI	No	4Q06	Yes	4Q07 or later	No	N/A	No	N/A	No	N/A
Cook 1*	85	900 (Phase 1 installed); Approx 2000 final	CCI	Yes	Phase 1 - 4Q06; Phase 2 - 2Q08	No	N/A	No	N/A	No	N/A	Yes - Water Management Pilot Plant. Specifics of changes TBD	N/A
Cook 2	85	Approx 2000	CCI	Yes	4Q07	No	N/A	No	N/A	No	N/A	Yes - Water Management Pilot Plant. Specifics of changes TBD	N/A
Crystal River Unit 3	86	1139	Enercon	Yes	4Q05	No	N/A	Yes	4Q09	Yes – cyclone separator modification	4Q07	No	N/A
Davis-Besse	50	1230	Enercon	Yes	2004	No	N/A	No	N/A	Yes (completed rather than planned). Included HPI pump modifications and cyclone separator modifications	2004	No	N/A
Diablo Canyon 1	700	3400	GE	Yes	2Q07	No	N/A	Yes	2Q07	No	N/A	No	N/A
Diablo Canyon 2*	700	3400	GE	Yes	1Q08	No	N/A	Yes	1Q08	No	N/A	No	N/A
Farley 1	100 to 200	3000	GE	No	1Q07	No	N/A	No	N/A	Yes, branch line resistance flow orifices	2Q07	No	N/A
Farley 2	100 to 200	3000	GE	No	4Q07	No	N/A	No	N/A	Yes - branch line resistance flow orifices	4Q07	No	N/A

Larger Screens + Less Debris = Lower Clogging Threat

Magnitude of GSI-191 Benefit

Source: NRC memo 08/20/2004

- **25 reactors very likely to have sump clogging for any size LOCA**
- **6 reactors very likely to have sump clogging for large and medium LOCA**
- **6 reactors very likely to have sump clogging for large LOCA and likely for medium LOCA**

Total averted cost: $172,000,000

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Generic Safety Issue 191, PWR Sump Performance

Michael Scott

Chief, Safety Issue Res. Branch

Office of Nuclear Reactor Regulation

September 29, 2010

Agenda

- NRC Staff Perspective
- Developments Since Last Brief
- Evaluation process for SRM items
- Approaches Considered
- Subjects of Disagreement
- Recommendations
- Conclusions

NRC Staff Perspective

- GSI-191 is a safety issue
- A small amount of debris can lead to sump clogging
- Large uncertainties in performance of sumps in "high-fiber" plants
- Tolerance of nuclear fuel to debris appears to be limited

NRC Staff Perspective (Cont'd)

- Small pipe breaks can be problems or even limiting
- Not all insulation is in zones of influence for breaks
- Some plants have resolved issues without insulation removal
- Dose impact not out of line with safety benefit

Developments Since Last Brief

- Progress continues – strainer test and evaluation methods found acceptable for 46 of 69 PWRS
- All vendors have now shown adequate test/evaluation protocol
- In-vessel "cross test" performed – unexpected results

Evaluations for SRM M100415

- Stakeholder interactions
- Evaluation teams for options
- Complex issues
- Several innovative approaches evaluated and/or adopted
- Met with CRGR

Approaches Considered

- Continue present resolution path
- No further actions needed
- Leak-before-break credit
- Risk-informed approach for some or all breaks

Approaches Considered (Continued)

- Split approach by leak size
- Resolve in-vessel effects separately

Potentially Viable Options

- Continue current approach, with or without firm schedule
- Risk-informed approach
- Leak-before-break (LBB) / General Design Criterion (GDC) 4

LBB Application to Sumps

- Large reduction in defense in depth
- Inconsistent with intent of GDC-4
- Potentially precedent-setting
- Primary water stress corrosion cracking
- Inconsistent with NRC/ACRS views on risk-informing ECCS

Staff-Recommended Approach

- Continue plant-specific issue resolution, demonstrated successful for many plants
- Risk-informed approach available to licensees to potentially mitigate evaluations for large breaks
- Different time limit for large and small breaks

Advantages of Staff Approach

- Maintains defense in depth
- Risk-informed
- Allows for proposed refinements
- Demonstrated effective issue resolution process

Conclusions

- GSI-191 approaches resolution, but remains a safety issue
- Staff recommends reasonably flexible, risk-informed approach
- In-vessel effects should be resolved as part of GSI-191 – uncertainty could affect closure schedule

Acronyms

- CRGR – Committee to Review Generic Requirements
- ECCS – emergency Core Cooling System
- GL – generic letter
- GSI – generic safety issue
- PWR – pressurized water reactor

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance"

Sanjoy Banerjee

Advisory Committee on Reactor Safeguards

September 29, 2010

September 17, 2010, ACRS Letter

- The staff has thoroughly reviewed three options to close GSI-191
- Option 1, which maintains the current holistic resolution process, and Option 2, which would develop risk-informed guidance that takes into account the lower probability of large-break loss of coolant accidents (LOCAs), are both acceptable, provided that a reasonable schedule for reaching resolution is adopted

September 17, 2010, ACRS Letter

- **Option 3, which would use GDC-4 to exclude debris generation during LOCAs for leak-before-break (LBB) qualified piping, should not be considered further**

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- **Although more information and analyses will become available in time, current experience suggests that doses for large scale replacement of insulation at the remaining plants are likely to be at or below the lower end of the range estimated by NEI**

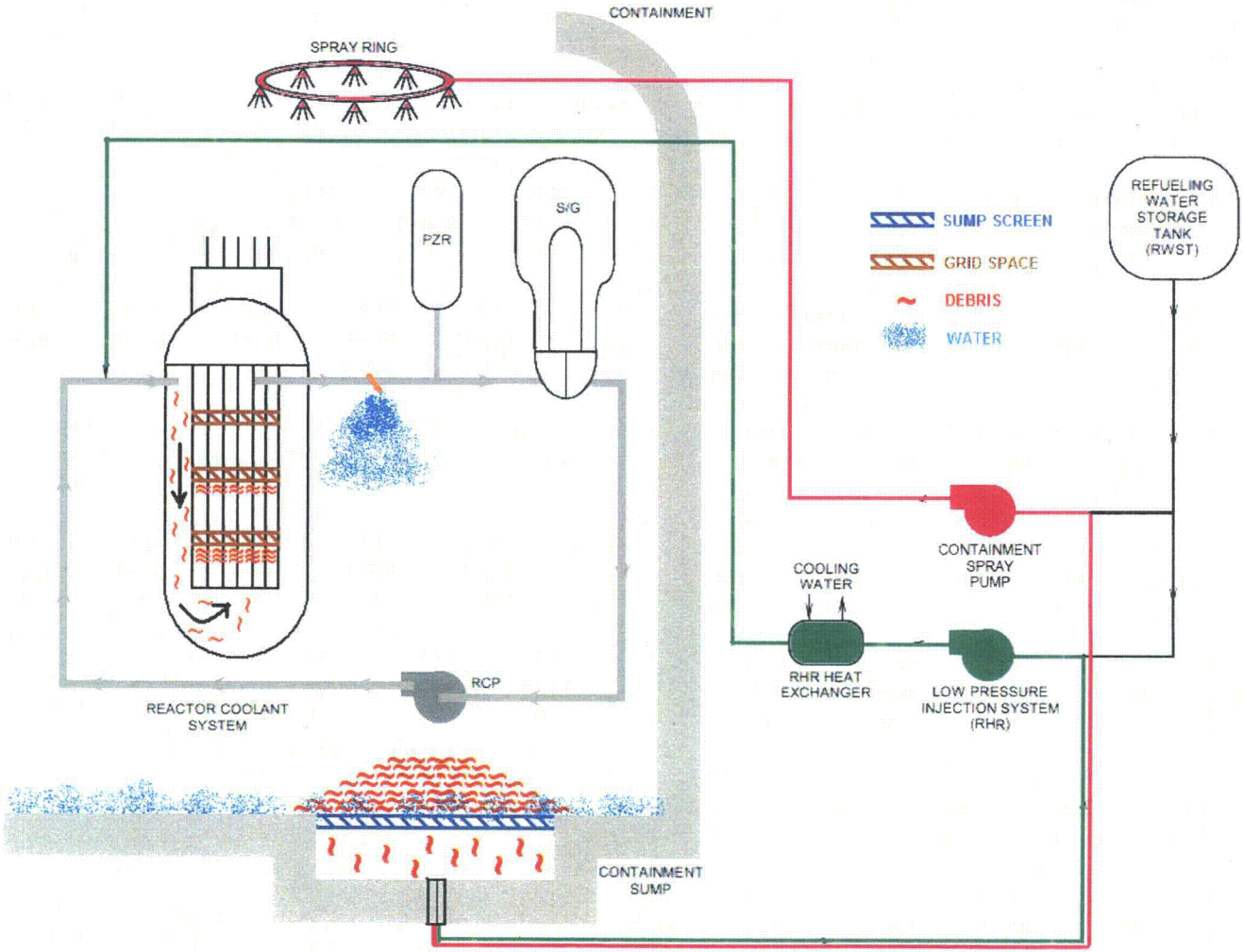
September 17, 2010, ACRS Letter

- ***In-vessel effects and sump screen performance are inextricably intertwined in determining the efficacy of long-term cooling. While it may delay ultimate resolution of GSI-191, we recommend against treatment of in-vessel effects as a separate generic issue***

Radiation Dose

- **NEI suggests dose estimates for insulation replacement in the range of 100 – 600 person-rem with an average of 200**
- **Current experience and more detailed estimates suggest doses will be at or below NEI's lower range**
  - ✓ **110 person-rem to replace insulation at 2 units of Beaver Valley**
  - ✓ **Estimated 81 person-rem per unit for large scale insulation replacement at STP**

Schematic of Long-term Recirculation System



Option 3

We recommend Option 3 not be considered

- **Inconsistent with defense-in-depth; could permit a LOCA to disable both the system that prevents core damage (ECCS) and the system that mitigates offsite releases (containment sprays)**
- **Does not maintain the capability to mitigate large break LOCAs contrary to Commission direction and ACRS recommendations for risk-informed changes to LOCA requirements (50.46(a))**

In-Vessel Effects

In-vessel effects should not be separated into another generic issue

- ***Sump screens and core act in concert to filter debris in the recirculation system***
- ***Large area screens may allow more flow but may allow more debris to pass through to the core***
- ***In-vessel blockage increases with increased debris loads that may pass through the screens***